NON-PUBLIC?: N
ACCESSION #: 9409160027
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Braidwood 1 PAGE: 1 OF 6

DOCKET NUMBER: 05000456

TITLE: Inadvertent Main Steam Line Isolation at Power due to
Equipment Failure
EVENT DATE: 08/11/94 LER #: 94-012-00 REPORT DATE: 09/09/94

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: M. Olson, Root Cause Team TELEPHONE: (815) 458-2801
x2028
COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: JG COMPONENT: ECBD MANUFACTURER: W120
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

At approximately 1028 a.m. on August 11, 1994, Braidwood Unit 1
experienced a spurious Train "A" Main Steam Line Isolation followed by an
automatic reactor trip from Low Low water level on the 1C Steam
Generator. Both Pressurizer PORV's
ycled as designed to relieve the RCS
pressure transient. Steam was released to the atmosphere via the steam
generator PORV's and Main Steam safety valves. Control Room Operators
stabilized the plant using the Emergency Operating Procedures. All
systems operated as designed following the MSIV closure to mitigate the
transient. During subsequent cooldown of the RCS a second Main Steam
Line Isolation signal was received which closed all four MSIV Bypass
valves. Troubleshooting determined the root cause to be a failure of the
Q7 output transistor on the A516 Safeguards Output Driver Board in Train
A of the Solid State Protection System. The failed A516 circuit board
was replaced and the Train "A" SSPS bimonthly surveillance was performed

satisfactorily.

END OF ABSTRACT

A. PLANT CONDITIONS PRIOR TO EVENT:

UNIT: Braidwood 1; EVENT DATE: August 11, 1994;
EVENT TIME: 1028;
MODE: 1; RX POWER: 100%;
RCS AB! TEMPERATURE/PRESSURE: NOT/NOP

B. DESCRIPTION OF EVENT:

On August 11, 1994, at approximately 1028 CDST, Unit 1 experienced a Main Steam Line Isolation followed by an automatic reactor trip on Low Low Steam Generator water level. Both of the Pressurizer PORV's cycled as designed after the MSIV closures to mitigate the primary system pressure transient. Steam was released to the atmosphere via the steam generator PORV's and safety valves. As a result of the high energy release from the steam generator PORV's and safety valves, a section of sheet metal siding was dislodged from the unit 1 containment roof walkway enclosure above the A/D safety valve room. The sheet metal fell on the turbine building roof with one section piercing a hole in the roof. An Unusual Event was declared at 1054 for "A failure that imparts significant energy to nearby structures." Control Room Operators stabilized the plant using the Emergency Operating Procedures and cooled the RCS to 557 degrees with the Steam Generator PORV's.

The appropriate NRC notification was made via the ENS phone system pursuant to 10CFR50.72(b)(2)(ii).

Interviews with Control Room Operators and review of the sequence of events recorder showed the cause of the reactor trip to be an automatic closure of all four Main Steam Isolation Valves. A subsequent walkdown of the Solid State Protection System (SSPS)revealed that the Main Steam Isolation was caused by the actuation of the SSPS Train "A" K616 and K623 output relays. Discussions were then held with senior plant management to develop a troubleshooting plan.

Troubleshooting began on shift 3 with efforts concentrating on the Main Steam Isolation actuation circuitry. monitoring equipment was installed to capture information that would pinpoint the exact component that caused the actuation. While monitoring the.actuation circuitry it was noticed that the A315 Universal Logic Circuit Board that processes the

Containment Pressure Hi-2 trip signals was exhibiting higher than normal noise on its output.

B. DESCRIPTION OF EVENT (continued):

By 1515, the Control Room Operators had completed Normal Operating Procedure 1BwOP MS-9, "Opening the Main Steam Isolation Valves", and a cooldown of the RCS utilizing the Steam Dumps was commenced. At 1555 another spurious Main Steam Isolation Signal was received on unit 1, closing all four Main Steam Isolation Bypass Valves being used for the cooldown. A review of the monitoring equipment installed allowed the elimination of several components that could be suspected as faulty but the exact cause of the spurious actuation signals could not yet be determined from the information gathered.

By 1637 the Control Room Operators had reset the Main Steam Isolation Signal and were reperforming 1BwOP MS-9 to reopen the MSIV Bypass valves for cooldown of the RCS via the Steam Dumps. The Steam Dumps were reopened at 1710. A second notification was made to the NRC via the ENS phone system pursuant to 10CFR50.72(b)(2)(ii) due to the second spurious MSI actuation signal.

Troubleshooting began at 2345 with the performance of 1BwOS 3.1.1-20, Unit 1 Train "A" SSPS Bimonthly Surveillance. An abnormality was discovered while testing the master relays. It was determined that a circuit on the A516 Safeguards Output Driver circuit board was not performing properly and was the cause for the abnormality. Upon further review it was found that this circuit on the A516 circuit board performs the MSI actuation. The A516 circuit board was replaced and the Train "A" SSPS bimonthly surveillance repeated. As a preventative measure, the A315 Universal Logic circuit board that had exhibited noise earlier was also replaced. 1BwOS 3.1.1-20 was completed satisfactorily at 0456 on 8/12/94 and Train "A" of SSPS returned to Operable condition.

This event is being reported under the requirements of 10CFR50.73(a)(2)(iv) - any event or condition that resulted in manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS).

C. CAUSE OF EVENT:

The cause of this reactor trip was Equipment Failure. The Main Steam Isolation signal resulted from Safeguards Output Board A516, circuit 2, whose output voltage had degraded sufficiently from the non-tripped state of 48 VDC to actuate the master relay K504 (MS Isolation). Subsequent bench testing identified the root cause to be a failure of the Q7 output transistor on the A516 Safeguards Output Driver Board.

D. SAFETY ANALYSIS:

This event had no effect on the safety of the plant or the public. There were no systems or components inoperable at the beginning of the event which contributed to the severity of the event.

In the Byron/Braidwood UFSAR Safety Analysis the Inadvertent Closure of the Main Steam Isolation Valves is addressed by the Turbine Trip analysis. The Turbine Trip event (UFSAR Section 15.2.3) is a Condition II event, a fault of moderate frequency. The limiting Turbine Trip is conservatively analyzed assuming no credit for either steam dump or steam generator PORV operation, and therefore is bounding and representative of the subject event.

The Turbine Trip analysis shows that DNBR does not decrease below its initial (previously steady state value) and therefore the event has no impact on core safety analysis. Four cases are evaluated to show the adequacy of the RCS pressure response which bound the full range of possible plant conditions. These cases reflect either maximum or minimum reactivity feedback in combination with either crediting pressurizer spray and PORV operation, or not crediting pressurizer spray and PORV operation. The most limiting cases are those in which pressurizer spray and PORV's are not credited. These cases are used to verify the adequacy of the Pressurizer System's relief valve capability and the peak pressure reaching the pressurizer safety valve setpoint. The plant pressure response for the subject event peaked at around the pressurizer PORV setpoint and is clearly bounded by the analyzed cases.

Though bounded by the more limiting cases described above, the subject event is most easily compared to the more representative cases where pressurizer spray and PORV's are available. Although the core was at beginning of life and not at maximum reactivity feedback conditions, the peak pressurizer pressure response was limited to approximately the pressurizer PORV setpoint. The only noted difference between the analyzed and actual plant response is that the plant tripped on Steam Generator Low Low Level in response to steam generator shrink. For the minimum feedback case, the reactor trip is reached in a similar time frame but the reactor trips on high pressurizer pressure due to a more severe pressurizer pressure increase. The reactor trip for the maximum reactivity feedback analysis is from Low Low Steam Generator Level but the analysis does not credit reactor trip until much later in the transient sequence.

D. SAFETY ANALYSIS (continued):

The plant response is bounded by the UFSAR analysis for Turbine Trip. The least limiting case, "With Pressurizer Pressure Control (Maximum reactivity feedback)", is most representative of the observed plant response. The Main Steam Safety Valves, Steam Generator PORV's, and Pressurizer PORV's all operated as designed to mitigate the transient event.

E. CORRECTIVE ACTIONS:

The A516 SSPS Safeguards Output Driver circuit board which performs the Train "A" Main Steam Isolation actuation was replaced. As a preventative measure, the A315 Universal Logic circuit board that had exhibited excessive noise during troubleshooting was also replaced. The unit 1 Train "A" SSPS bimonthly surveillance, 1BwOS 3.1.1-20, was then performed satisfactorily.

A walkdown of all Unit 1 Main Steam piping in the main Steam tunnel, the Main Turbine Building, and in Containment was conducted post-trip. In addition, the steam relief valve rooms were inspected in detail, including the supports on the relief valves. All equipment visually inspected was acceptable,with no indication of damage from the event.

The Site Engineering Department visually inspected the main steel framing for the Containment walkway enclosure and determined that no structural damage had occurred. The integrity of the walkway structure was not affected. All sections of the sheet metal siding which were blown off the walkway structure have been accounted for and have been placed in storage for future evaluation. Site Engineering is in the process of developing design details which would preclude the sheet metal siding from being blown off in the future. An Engineering Request has been written to follow this design study to completion.

F. PREVIOUS OCCURRENCES:

There have been three previous occurrences of spurious ESF actuations at Braidwood Station due to problems in the SSPS circuitry. Previous corrective actions addressed the root and contributing causes.

LER 90-018; Spurious Train B SSPS actuations due to component failure, personnel error, and component interface design deficiency - 2 reactor trips, 2 safety injections. Root cause attributed to an intermittent failure in the output function of SSPS.

F. PREVIOUS OCCURRENCES (continued):

LER 1-92-013; Feedwater Isolation due to spurious actuation of the Train B Feedwater Isolation circuitry. Conditions were not repeatable during SSPS Bimonthly surve
llance, no root cause identifiable.

LER 1-93-001; Reactor Trip on Low Reactor Coolant Flow signal during performance of 1BwVS 3.1.1-6 due to logic card failure. Root cause attributed to the failure of a logic card in Train B of SSPS.

A limited review of NPRDS was conducted for other industry events resulting in a reactor trip or unit shutdown where the initiating event was a spurious actuation. This review was conducted to identify areas or items that have the potential to create problems where preventative measures could be taken at Braidwood Station.

Many of these events were due to random electronic component failures that did not show detectable degradation prior to failure. As a result, there are no identified items that could be implemented at Braidwood to preclude additional spurious actuation signals of this type.

G. COMPONENT FAILURE DATA:

MANUFACTURER NOMENCLATURE MODEL MFG PART NO.

RCA Transistor N/A 2N2405

ATTACHMENT TO 9409160027 PAGE 1 OF 1

Commonwealth Edison

Braidwood Nuclear Power Station
Route #1, Box 84
Braceville, Illinois 60407
Telephone 815/458-2801

September 9, 1994
BW/94-0149

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Dear Sir:

The enclosed Licensee Event Report from Braidwood Generating Station is being transmitted in accordance with the requirement of 10CFR50.73 (a)(2)(i)(v), which requires a 30-day written report.

This report is number 94-012-00, Docket No. 50-456.

K. L. Kofron
Station Manager
Braidwood Nuclear Station

KLK/CP/dla
o:corresp\zcbw94

Encl: Licensee Event Report
No. 456/94-012-00

cc: NRC Region III Administrator
NRC Resident Inspector
INPO Record Center
Illinois Dept. of Nuclear Safety

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